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September 24, 2003

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
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SUBJECT: Entergy Nuclear Operations, Inc.
Pilgrim Nuclear Power Station
Docket No. 50-293
License No. DPR-35

10 CFR 50.59 Report - Changes, Tests, and Experiments Performed at Pilgrim Nuclear Power Station, for the period July 1, 2001 to June 30, 2003.

LETTER NUMBER: 2.03.110

In accordance with 10 CFR 50.59(d)(2), find enclosed the report of the changes, tests, and experiments evaluated in accordance with 10 CFR 50.59 at Pilgrim Nuclear Power Station for the period of July 1, 2001 through June 30, 2003.

If you have any questions or require additional information, please contact Mr. Bryan Ford, Licensing Manager, at (508) 830-8403.

Sincerely,

W.J. Riggs

Enclosure: 10 CFR 50.59 Report of Changes, Tests, and Experiments (9 pages)

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ENCLOSURE

10 CFR 50.59 Report of Changes, Tests, and Experiments

10 CFR 50.59 REPORT OF CHANGES, TESTS, AND EXPERIMENTS

The following provides a summary evaluation report of changes, tests, and experiments conducted at Pilgrim Nuclear Power Station for the period July 1, 2001 to June 30, 2003. This summary report is provided pursuant to 10 CFR 50.59(d)(2).

Evaluation Number 3384, Revision 0

Description: This modification supported the addition of Extended Test System (ETS) and Augmented Offgas (AOG) inputs to the Kaye computer. The new ETS inputs to Kaye replaced the ETS computer and data acquisition system because it was outdated, tripped frequently, and was labor intensive to maintain. Additionally, AOG cooler Condenser and Offgas flow trending information was needed to monitor AOG performance. These two inputs replaced a temporary modification and an abandoned multiplexer. They provided the Control Room with more reliable equipment performance.

Summary: The addition of new inputs to the Kaye computer does not introduce any new failure modes or accident initiators. The ETS system and the AOG systems are not functionally impacted by this change. Control and Alarm functions for both ETS and AOG are not affected. This new Kaye computer equipment has the same function as existing equipment, which has been tested, and in service at PNPS for approximately 6 years. The guidance contained in EPRI TR-102323-R1 and TR-102348 for digital equipment upgrades was followed for this modification. All failure modes are the same as the existing Kaye computer and Kaye Netpac equipment.

Evaluation Number 3385, Revision 0

Description: This modification increased the time-delay setting for emergency diesel generator (EDG) timers 162A-509 and 162A-609 in order to increase available time for load center breakers B102 and B601 to pre-charge. This assures that ample time is provided for breaker pre-charge under a design basis loss of coolant accident (LOCA) or Turbine Trip coincident with degraded voltage conditions, thereby ensuring the 480V breakers perform their function as designed.

Summary: Although the modification increased the time delay for re-powering emergency buses from 15.7 seconds to 17.8 seconds, the new time is still below the maximum value of 19.7 seconds assumed in the analysis of record. Effects on electrical coordination schemes including pump start sequences and valve opening/closing sequences were re-analyzed. The re-analysis demonstrated the modification eliminates potential problems related to improper/inadequate pre-charge of load center breakers and that there is no adverse impact on the electrical coordination or pump/valve operating sequences. Furthermore, this modification was evaluated against LOCA analysis, NEDC-31852P, Rev. 1, as referenced in the Pilgrim Updated Final Safety Analysis Report (UFSAR). This analysis bounds the increase in timing for EDGs to restore power following a loss of offsite power (LOOP) condition.

Evaluation Number 3386, Revision 0

Description: This evaluation supported the application of a freeze seal to 2" line feeding Chemistry Laboratory industrial safety shower to permit replacement of non-functioning isolation valves. Degraded valves in the line feeding the Chemistry Lab safety shower restricted the flow of water. A freeze seal was applied to the line upstream of the valves to provide isolation so that the valves could be replaced. The industrial safety shower uses potable city water.

Summary: The approved freeze seal procedure provided adequate controls to control the evolution and preclude failures in conjunction with maintaining plant configuration within technical specification requirements. The evaluation concluded that application of this particular freeze seal, under specific procedural conditions, on the domestic city water system did not increase the probability of occurrence or consequences of a previously analyzed accident or equipment malfunction, nor did it create the possibility of a new accident or malfunction of equipment important to safety.

Evaluation Number 3387, Revision, 0

Description: This evaluation number was not used.

Summary: Not Applicable.

Evaluation Number 3388, Revision 0

Description: This change replaced an existing safety related relay in automatic transfer switch Y-10 with two safety related relays. Y-10 is an automatic transfer switch that provides power from either safety train to 125 VDC Distribution Panel D6. This change was necessary because a single replacement relay with the necessary contacts could not be found. The two replacement relays are safety-related and are installed in parallel to perform the identical functions as the original.

Summary: The safety function of 125 VDC Distribution Panel D6 was not impacted. Although an additional relay was added, the Y10 circuitry will operate using the same under voltage signals and the new relay contacts will energize the same circuits and associated relays as the original relay. The new relays and circuit design maintain both automatic transfer capability and separation of the safety trains. Overall reliability is improved because the relays are a newer design and meet more stringent test standards. The higher current draw and weight of the new relays was evaluated and found to have a negligible impact.

Evaluation Number 3389, Revision 0

Description: This change incorporated Emergency Procedure / Severe Accident guidelines (EPG/SAG) developed from the BWROG generic document EPG/SAG Rev. 2 into the Emergency Operating Procedures (EOPs). The previous (EOPs) were based on EPG Rev. 4 issued by the BWROG and approved by the NRC. EPG/SAG Rev. 0 and Rev. 1 were not implemented at PNPS. The major changes between Rev. 4 of the EPG and EPG/SAG Rev. 2 are:

- Revision of the definition of adequate core cooling,
- Inclusion of the EPG changes to address ATWS/Stability events,
- Changes to include Severe Accident Management Guidance.

Summary: The EOPs, based on EPG Rev. 4, were reviewed and approved for use at PNPS by the NRC (NRC Letter, "Pilgrim Procedures Generation Package – Safety Evaluation," dated June 6, 1988). The ATWS/Stability changes included in EPG/SAG Rev. 2 were previously approved as a modification to EPG Rev. 4 (NRC Letter, "Staff Review of Modifications to Revision 4 of the Boiling Water Reactor (BWR) Emergency Procedure Guidelines," dated June 24, 1996). Use of EOPs and SAGs is an integral part of the PNPS Emergency Response Organization during an accident or transient and does not affect accident initiators. Because the revisions to the EOPs are bounded by existing safety evaluations and the PNPS licensing basis, they do not increase the consequences of events evaluated in the UFSAR. The EOPs/SAGs address conditions that are both within and beyond the PNPS design and licensing basis and provide the best direction for all mechanistically possible events and therefore minimize the impact on the health and safety of the public. Operation of equipment beyond design limits is only authorized when conditions are clearly beyond the PNPS design and licensing basis.

Evaluation Number 3390, Revision 0

Description: This change provided level indication for the Station Blackout Diesel Generator (SBODG) Fuel Oil Tanks and replaced the tank leak detection system with components compatible with the level monitoring system. In addition to the SBODG, the fuel oil tanks also supply a portion of the required inventory for EDGs.

Summary: There was no direct or indirect connection between the level monitoring system and the functions provided by the SBODG and EDGs. The new components cannot initiate an accident and their failure has no impact on components that affect the consequences of any accident described in Chapter 14 of the UFSAR.

Evaluation Number 3391, Revision 0

Description: The design basis performance of the Augmented Offgas (AOG) System, as described in UFSAR and AOG system surveillance procedures, was updated to provide more appropriate design values for noble gas hold-up times and Offgas flow rates and to provide a better description of the parameters used to define the acceptable performance envelope.

Summary: This evaluation resulted in an updated AOG system design basis for Krypton and Xenon holdup times, Curie Reduction Factor, site boundary dose, noble gas release at the Steam Jet Air Ejectors (SJAE), noble gas input to charcoal, and condenser air in-leakage. These new values were based on more realistic rates of condenser air in-leakage and the resulting AOG system flow rate. This evaluation also established a range within which the charcoal vault temperature may vary, as long as AOG holdup times remain within the established envelope. This evaluation included changes to the UFSAR documenting the new system design bases and a figure defining the established envelope for acceptable AOG operation. The evaluation concluded that the AOG system potential failure modes are unaffected by the updated system design basis and operational performance. Also, the updated AOG system design basis provides full compliance with 10 CFR 20 for radiation dose rates to members of the public at the site boundary and the corresponding limits given in the Offsite Dose Calculation Manual (ODCM).

Evaluation Number 3392, Revision 0

Description: Calculation PNPS-1-ERHS-XIII.BB was performed to estimate the radiological consequences of a LOCA at 102% of 1998 MWth. This calculation also updated the environmental releases by inclusion of ECCS and MSIV leakage dose contributions along with the drywell leakage dose. PNPS specific meteorological factors determined in accordance with NRC approved models were also applied.

Summary: The estimated offsite consequences of a postulated design basis LOCA were re-evaluated to be in accordance with regulatory guidance in regards to power level, leakage pathways, and PNPS-specific atmospheric dispersion factors. The method for calculating offsite doses was modified to adjust the power level to 102% of core thermal power and to include radioactivity releases due to ECCS leakage and MSIV leakage, as well as releases due to drywell leakage, which were not previously considered. Other input parameter values were updated to reflect PNPS-specific values. The calculated offsite doses at the exclusion area boundary and the low population zone were significantly below the regulatory limits given in 10 CFR 100; therefore, all systems, structures, and components meet their intended safety functions.

Evaluation Number 3393, Revision 0

Description: Temporary Modification 02-09, "Reactor Level Reference Leg Backfill System" was implemented to affect the compensatory measures identified in Engineering Evaluation 02-011 to ensure the continued operability of the "B" ECCS reactor water level instrumentation by placing the reference leg backfill system into continuous service. This was done to prevent the migration of non-condensible gases into the reference leg of the "B" ECCS reactor water level instrumentation thereby preventing the possibility of water level "notching" when the plant is shut down and depressurized.

Summary: The compensatory measures proposed in Engineering Evaluation 02-011 could only affect the instruments connected to the condensing chamber 12B (train "B" ECCS reactor water level instrumentation) and would not affect the redundant "A" train or any other system, structure, or component. A failure modes and effects analysis was performed to ensure that the compensatory measures rendered the "B" ECCS reactor water level instrumentation operable to meet its design objectives within the bounds of the existing plant safety analysis. Furthermore, in satisfying the separation requirements for the redundant trains, the implementing procedures affected by this change provided the necessary administrative controls to preclude cross connecting the trains. The implementing procedures also recognized that a Limiting Condition of Operation is entered if the trains are both connected simultaneously to the backfill system.

Evaluation Number 3394, Revision 0

Description: This evaluation supported a change to a Temporary Procedure TP-01-59 for EDG post work testing after an EDG governor replacement and determined that a license amendment was not required. The proposed change involved load acceptance and load reject testing of an EDG following replacement of the EDG governor while the plant was on-line. The proposed post work testing scope included starting and stopping the largest single load on the safety bus (i.e., a core spray pump) while the safety bus was loaded on the EDG. This test was to prove that the EDG was operable. During review of License Amendment #179 (the EDG 14 day out of service time change), the NRC was concerned about the scope of work for an on-line EDG overhaul, and whether or not the scope of work included post work testing of the replacement governor.

Summary: Review of 'Requests for Additional Information' and responses related to Amendment #179 found that PNPS stated "some form" of load reject testing would likely be performed should the need for governor replacement arise. The concern about governor testing was in the context of potential damage to the safety busses and loads during post work testing. As explained in correspondence for Amendment #179 the maintenance process included reviews to ensure adequate testing was performed. Testing after governor tuning has already been performed which affirms the conclusion that the test is safe and the maintenance process reviews minimized the potential for inadequate testing. The partial load testing contained in the test procedure was appropriate. The proposed method of testing was consistent with the way the EDG was assumed to operate during design basis accidents.

Evaluation Number 3395, Revision 0

Description: This evaluation supported a design change to increase the design basis discharge capacity of the safety relief valves (SRV). The design basis SRV discharge flow capacity was increased from 800,000 lbs/hr per valve at a set pressure of 1095 psig to 862,125 lbs/hr per valve at a set pressure of 1080 psig. The increase in SRV capacity was achieved by physically enlarging the SRV throat diameter by one-eighth inch. The SRV set pressure, and normal operating pressure, and temperature remain unchanged. The SRV throat diameter modification, which did not affect the Technical Specifications related to SRVs, was made to support the increase in the reactor thermal power limit (TPO uprate) license amendment. The TPO uprate increased main steam system flow rates, which necessitated a comparable increase in SRV capacity. The TPO uprate License Amendment No. 201 increased the licensed thermal power from 1998 MWt to 2028 MWt. The UFSAR and Technical Specification Bases will be updated to reflect the implemented changes including the modification to the SRV throat diameter. The piping analysis to support the SRV throat diameter modification used a methodology that was approved by the NRC as part of the TPO License Amendment No. 201.

Summary: The SRV throat diameter modification added margin to the (pre-TPO) design. Licensing basis requirements impose a minimum discharge capacity on the SRVs. Adding capacity above minimum requirements was conservative in the context of the pre-uprate license. The design basis discharge flow rate for this modification was selected in anticipation of satisfying requirements of a higher reactor thermal power level and associated increase in main steam system flow rate for the TPO uprate license amendment. A detailed thermal-hydraulic analysis was performed that used more severe SRV discharge flow parameters than the proposed changes, which demonstrated there would be no significant change to torus pressure and therefore no challenge to the containment.

Evaluation Number 3396, Revision 0

Description: This evaluation supported design change, which added permanent circuitry and key-locked electrical switches to the Reactor Protection System (RPS) and Primary Containment Isolation System (PCIS). Certain interlocks and initiation logic must sometimes be bypassed to permit execution of Emergency Operating Procedure (EOP) or Severe Accident Guideline (SAG) steps. Approved procedures provide methods for performing these bypasses under the control of the EOP/SAGs as needed to implement mitigation strategies. The modification installed permanent circuits with key-locked controls to replace the current method of separate jumper wires. The components added provided the same function for event mitigation as the separate jumper wires and will still be administratively controlled by the EOP/SAGs implementing procedures. The use of this modification minimized the potential for human error in a stressed situation if the jumpers were to be manually installed.

Summary: The design change did not change the design function or intent of the RPS or PCIS components or system. Evaluations for the use and control of interlock bypass have previously been reviewed and approved by NRC that allowed for bypass of these functions under specific conditions controlled by EOP/SAG implementing procedures. Those considerations and administrative conditions remain unchanged by addition of the switch components, which are intrinsically conservative in that they eliminate multiple occasions for human error while maintaining the same level of reliability as required in the existing circuit and components criteria.

Evaluation Number 3397, Revision 0

Description: This evaluation supported Reload 14 / Cycle 15 core design, which replaced 164 irradiated fuel assemblies with 164 fresh GE14C fuel assemblies (maximum bundle average enrichment of 3.98 weight percent). GE14C fuel was first introduced in the PNPS core for Reload 13 / Cycle 14 and is a 10 x 10 lattice design. The Cycle 15 core design was based on an uprated power level of 2028 MWth, which represents a 1.5% increase from the current licensed power level. The Technical Safety Analysis Report (TSAR) associated with the power uprate license amendment specified that some aspects of the plant design and limits be addressed in the first reload analysis for the power uprate. This evaluation in conjunction with the license amendment submittal for this power uprate addressed the full scope of issues requiring evaluation per the TSAR. Furthermore, a complete 10 CFR 50.46 LOCA analysis was performed for both fuels currently in use (GE11 and GE14). This updated LOCA analysis explicitly evaluated additional electrical distribution system time delays for their estimated impact on Peak Clad Temperature (PCT). Also, this updated analysis removed all outstanding errors against the analysis for GE11 fuel based on the use of the current corrected set of computer codes associated with the SAFER/GESTR-LOCA methodology. Therefore, all known 10 CFR 50.46 LOCA analysis errors are explicitly accounted for in the analysis.

Summary: This evaluation concluded that the Cycle 15 core design including the 10 CFR 50.46 analyses satisfied all licensing criteria from the PNPS UFSAR and the General Electric Standard Application for Reactor Fuel (GESTAR). This evaluation did not establish the acceptability of the change to the licensed thermal power and changes to the power flow map initiated by the TSAR. These changes were subject to approval of the license amendment submittal for the power uprate.

Evaluation Number 3397, Revision 1

Description: Revision 1 of 50.59 evaluation No. 3397 was performed due to changes in the final core reload pattern. There was a change to the reference-loading pattern specified in the Fuel Loading Plan. Global Nuclear Fuels (GNF) performed analysis of the revised reference-loading pattern. The Reload 14 / Cycle 15 core design replaced 164 irradiated fuel assemblies with 164 fresh GE14C fuel assemblies (maximum bundle average enrichment of 3.98 weight percent). GE14C fuel was first introduced in the PNPS core for Reload 13 / Cycle 14 and is a 10 x 10 lattice design. The Cycle 15 core design was based on an uprated power level of 2028 MWth, which represents a 1.5% increase from the current licensed power level. The TSAR specified that some aspects of the plant design and limits be addressed in the first reload analysis for the power uprate. This evaluation in conjunction with the license amendment submittal for this power uprate addressed the full scope of issues requiring evaluation per the TSAR. Furthermore, a complete 10 CFR 50.46 LOCA analysis was performed for both fuels in use currently (GE11 and GE14). This updated LOCA analysis explicitly evaluated additional electrical distribution system time delays for their estimated impact on PCT. Also, this updated analysis removed all outstanding errors against the analysis for GE11 fuel based on the use of the current corrected set of computer codes associated with the SAFER/GESTR-LOCA methodology. Therefore, all known 10 CFR 50.46 LOCA analysis errors were explicitly accounted for in the analysis.

Summary: GNF performed an analysis of the change to the reference-loading plan and concluded that criteria as stated in section 3.4 of GSTAR were met. The final fuel-loading plan met all criteria of GSTAR and was acceptable because licensing calculations performed on the reference core remained applicable. This evaluation concluded that the Cycle 15 core design, including the 10 CFR 50.46 analyses, satisfied all licensing criteria from the PNPS UFSAR and the General Electric Standard Application for Reactor Fuel (GESTAR). This evaluation did not establish the acceptability of the change to the licensed thermal power and changes to the power flow map initiated by the TSAR. These changes were subject to approval of the license amendment submittal for the power uprate.

Evaluation Number 3398, Revision 0

Description: This evaluation supported a definition of 'Operation with the Potential to Drain the Reactor Vessel (OPDRV) in PNPS Procedures and UFSAR. This term is used in PNPS Technical Specifications (TS) [3.7.B.1.a, 3.7.B.2.a, 3.7.C.1, 3.7.C.2.c, bases 3 / 4.7C] but not defined in TS.

Summary: The definition as evaluated was constructed based on research other plant's licensing documents and NRC memorandum dated 1994 concerning a Fermi Plant event. The NRC memo stated that there was no consistent definition for OPDRV throughout the industry and offered a proposed definition. The evaluation concluded that the addition of the OPRDV definition did not alter prior evaluations or assumptions of the UFSAR and introduced no new failures or failures of a different type than evaluated in the UFSAR. The evaluation therefore concluded that prior NRC approval was not required to incorporate the definition in the UFSAR.